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Nuclear Power Plant
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New York Power Authority

Robert J. Barrett
Site Executive Officer

July 5, 2000
IPN-00-052

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Licensee Event Report # 2000-007-00
Automatic Reactor Trip After Turbine Trip Due to Low Level in Steam Generator 31 As a Result of Personnel Error During Transfer of Feedwater Control

Dear Sir:

The attached Licensee Event Report (LER) 2000-007-00 is hereby submitted as required by 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73 (a)(2)(iv) for a condition recorded in the New York Power Authority's (NYPA) corrective action process as Deviation Event Report DER 00-01343.

NYPA is making no new commitments in this LER.

Very truly yours,

Robert J. Barrett
Site Executive Officer
Indian Point 3 Nuclear Power Plant

cc: See next page

IE22

cc: Mr. Hubert J. Miller
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U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point 3	05000286	1 OF 5

TITLE (4)

Automatic Reactor Trip After Turbine Trip Due to Low Level in Steam Generator 31 As a Result of Personnel Error During Transfer of Feedwater Control

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	04	2000	2000	007	00	07	05	2000		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	22

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	✓	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A	
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
John Janicki, Senior Operations Engineer	(914) 736-8813

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	✓ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 4, 2000, an automatic reactor trip (RT) occurred during plant startup with the main turbine generator tied to the grid and reactor power being raised to 30%. At approximately 22% reactor power, control of feedwater (FW) to the steam generators (SG) was being transferred from bypass feedwater flow to main FW flow by manually adjusting FW regulating valves (FRV). When adjustment to main boiler FW pump speed was made by a licensed operator, FW flow increased and the level in the 33 SG reached the high-high level setpoint initiating a turbine trip (TT). The TT caused a rapid decrease in SG water level due to shrink. When the 31 SG reached the low-low level setpoint an automatic RT was initiated. Required equipment functioned as designed. The power operated relief valves (PORVs) and pressurizer safety valves did not lift due to the transient. Auxiliary feedwater actuated per design as a result of the RT. The plant was stabilized in the hot shutdown condition. The RT was caused by a cognitive personnel error by a licensed operator who failed to adequately control FW pump speed during direction of an operator trainee who was controlling FRVs. Significant corrective actions taken include removal of responsible licensed operators from shift duty, preparation of an action plan for assessment of operations personnel performance and identification of improvements/recommendations. The event had no effect on public health and safety.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Note: The Energy Industry identification system Codes are identified within the brackets { }

DESCRIPTION OF EVENT

On June 4, 2000, at approximately 0554 hours, the reactor protection system {JC} initiated an automatic reactor trip (RT) during startup from a maintenance outage. At the time of the event, plant startup was in progress, the main turbine generator {TB} was tied to the grid and reactor power was being raised to 30%. At approximately 22% reactor power, the control of feedwater (FW) {SJ} to the steam generators (SG) {AB} was being manually controlled by an authorized operator trainee under the direct control of a qualified licensed operator. During the transfer from bypass FW regulating valves (FRV) {FCV} to main FRV, an adjustment to main boiler FW pump (MBFP) {P} speed was made by the licensed operator. FW flow increased and the level in the 33 SG reached the high-high level setpoint (75%) and a turbine trip (TT) {JD} was initiated. The turbine tripped automatically at 0552 hours, causing a rapid decrease in SG water level due to shrink. Subsequent to TT, shrinkage in the water level for the 31 SG reached the low-low level setpoint (8%) and a RT was initiated at approximately 0554 hours.

Control Room {NA} Operators observed the Turbine Trip First Out Annunciator {ANN}, "33 Steam Generator High Level," then the rod bottom lights and the Reactor Trip First Out Annunciator, "31 Steam Generator Low Level." Operators subsequently entered Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection," initiated event recovery, stabilized the plant and transitioned to the hot shutdown condition. Plant protective equipment operated as expected in response to the event; all reactor trip {JC} breakers {BKR} opened, all control rods {AA} fully inserted, main FW isolated {SJ}, and auxiliary feedwater {AFW} {BA} automatically started. No safety injection actuation {JE} occurred nor was one required. Reactor Coolant System {AB} pressure remained below the setpoint for pressurizer PORVs and Code Safety Valve {RV} actuation. Offsite power {EB} remained available during the event.

Prior to the event a composite operations crew was on shift in the control room (CR) composed of a shift manager (SM), control room supervisor (CRS), a reactor operator (RO), balance of plant (BOP) RO, an RO for the turbine, a shift technical advisor (STA), and three (3) operators in-training. The three operator trainees were assigned to work at the controls under instruction. The three licensed operators on duty were each instructing an operator trainee; one on the reactor, one on the FW system, and one on turbine control. The SM briefed the crew on the startup which included overall control of trainees. There was no specific brief on the transfer of FW control prior to performing the activity. At approximately 0500 hours, operator trainees performed activities associated with ascension to approximately 12% reactor power. At 0507 hours, operator trainees under supervision performed an escalation to 20% power. The SG level control trainee was under direct control of the BOP RO assigned to the FW system. Initially the main and bypass FRVs are closed, the MBFP started and AFW flow is transferred to FW low flow bypass feed by throttling open the bypass FRVs while closing the AFW FRVs.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

The delivery of FW to the SGs and the control of SG water level is regulated by FRV BFD-FCV-417, 427, 437, and 447. At low power conditions, FW flow is controlled by a low flow bypass around each FRV. Each low flow bypass has a low flow bypass FRV (BFD-FCV-417L, 427L, 437L, and 447L) which is capable of remote manual control from the CR. The position of the FRVs and the speed of the MBFPs are normally controlled by the SG level control system (SGLCS) to automatically maintain the water inventory in each SG. During startup/low power conditions the SGLCS is switched to manual and the position of the FRVs and the speed of the MBFPs are manually controlled by an operator. The SGLCS is a subsystem of the reactor control and protection system (JC).

In accordance with the FW system operating procedure (SOP-FW-1), when the bypass valves are at approximately 75% open and SG level stable at approximately 200 MWe, the transfer from low flow bypass feed to main FW feed should begin. At 0529 hours, the RO assigned to FW control misconstrued the CRS's reading of a procedural step as permission to transfer from low FW flow to main FW flow. The SG water level control trainee, under direction of the RO assigned to FW, began shifting from low FW flow (bypass) to main FW flow by manipulating the associated FRVs. At 0541 hours, during transfer of FW flow control by the SG water level control trainee, the RO assigned to FW noticed that water level in SG 31 was increasing. SG 31 flow was decreased at approximately 0549 hours, and at 0550 hours SG 31 level was trending down. During this time the FW RO failed to monitor MBFP status adequately and when the FW RO's attention returned to the MBFP it was out-of-band low on differential pressure. At the time the MBFP status was checked the water level in SG 33 was high. The FW RO increased MBFP speed without notifying the trainee or the CRS. The increase in MBFP output accelerated the rate of increase of water level in SG 33. At approximately 0551 hours, the FW RO through direction of the SG level control trainee attempted to close the FRVs to prevent reaching the SG high level trip. At approximately 0552 hours, the water level in SG 33 reached the high-high level setpoint (75%) and satisfied the TT logic of 2 out of 3 level transmitters and a TT was initiated. The TT resulted in SG level shrink which caused 2 out of 3 water level transmitters in SG 31 to reach their low-low level setpoint (8%) initiating an automatic RT at approximately 0554 hours. An anticipatory RT on TT did not occur because reactor thermal power was below the Reactor Protection System (RPS) permissive (P-8) setpoint. A post transient review was performed for the plant trip.

CAUSE OF EVENT

The direct cause of the RT was due to satisfying the reactor protection system (RPS) logic of two out of three (2/3) low-low SG water level (8%) in SG 31. The low-low water level in SG 31 was due to a TT and resultant shrink effect on SG water level. The TT was initiated as a result of satisfying the trip logic of two out of three (2/3) high-high SG water level (75%) in SG 33. The high water level in SG 33 was due to a cognitive personnel error by an licensed operator (FW RO) who failed to adequately control FW pump differential pressure during direct instruction control of an operator trainee controlling FRVs. The late recognition of FW pump differential pressure and pump adjustment without proper consideration of FRV position and SG level resulted in overfeeding the SGs.

Contributing factors were inadequate verbal communication, poor work practices and supervisory methods.

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

The composite shift crew had not trained or worked together prior to this event and several watch crew members had limited startup experience at their assignments. The SM performed an overall briefing of startup, but there was no specific briefing on the actions to perform the swapover from bypass FRV to main FRV. The SM was in the CR when operators were transitioning from bypass to main FRVs but was not notified of this evolution nor that it was to be performed by a operator trainee. The control of FW at low power and the transition from manual to automatic control was not made an infrequently performed evolution. The SG water level control trainee had not trained on or performed the task of SG level control on the simulator before. The FW RO did not communicate to the CRS or the trainee that he was increasing MBFP speed during FRV swapover. Neither the CRS nor the SM actively intervened into the actions of the FW RO.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Authority's corrective action program to address the cause of this event:

- The personnel directly involved in the personnel error (FW RO, CRS, SM) were removed from shift duty.
- An action plan (AP) was developed to assess operations personnel performance, identify improvements and provide recommendations for implementation. The AP will require personnel removed from the watch to develop a monitored self improvement plan (MSIP). Some of the factors that the AP includes for evaluation or consideration are:
  - Pre-job briefings prior to transferring from bypass FRVs to main FRVs.
  - Administrative controls regarding the transition from low flow bypass to main FRVs (e.g., one FRV at a time).
  - Restricting the number of trainees under instruction in the CR.
  - Revision to Administrative Procedure AP-19.1,"Infrequently Performed Tests and Evolutions."
  - Evaluation of the adequacy of operator training in supervising trainees under instruction.
  - Review/evaluation of crew composition when in a "non-standard alignment.

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ANALYSIS OF EVENT

The event is reportable under 10 CFR 50.73 (a) (2) (iv). The licensee shall report any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

This event meets the reporting criteria because an automatic RT occurred as a result of satisfying the reactor protection system logic for a RT. Two out of three low-low water level in SG 31 generated an automatic RT signal. AFW, an Engineered Safety Feature (ESF), automatically started in response to the RT. Operations made a four hour non-emergency notification to the NRC at 0642 hours, for a RPS actuation in accordance with 10 CFR 50.72 (b)(2)(ii), (ENS Log No. 37054).

A review of the past two years of Licensee Event Reports (LER) for events that involved RTs identified the following: LER 98-006, LER 1999-010, and LER 1999-003-01. LER 1999-003-01 was due to inadequate planning and change management for a temporary modification. The other LERs were a result of equipment failure. There were no previous events identified during this time that were due to inadequate operator control of FW flow which caused a RT. Corrective action for these LERs would not have prevented this event because they corrected different causes.

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because the low SG water level was due to SG shrink as a result of a TT caused by operator error not equipment failure or an accident. Following the RT, the plant was maintained stable in the hot shutdown condition. There were no potential safety consequences of the event under reasonable and credible alternative conditions. During full power operation the FRVs and MBFP speed control are under automatic control of the SGLCS preventing the operator error of this event. The protection systems are designed to fail into a safe state. The FRVs are designed to fail in the safe (closed) position on loss of instrument air or electrical power. For a RT, the FRVs open if the average reactor coolant temperature exceeds a set temperature but automatically close on low reactor coolant temperature combined with a RT signal. Each RT circuit was designed so that a trip occurs when the circuit is de-energized. RT is a fail safe condition. The FRVs automatically close on a safety injection signal. Plant protective functions were available and did actuate as designed. The low-low SG water level trip is to protect the reactor from a loss of heat sink. The function of the high-high SG water level trip is to terminate feedwater flow for excessive feedwater events; however it is not required for core protection.

This event is not a safety system functional failure (SSFF) per the guidelines of NEI 99-02, Revision 0. Safety systems in response to this event were capable of performing their safety function and they performed as expected.